10CFR50.73

Virginia Electric and Power Company
Surry Power Station
5570 Hog Island Road
Surry, Virginia 23883

March 26, 2003

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Serial No.: 03-197
SPS: BAG/TJN R2
Docket No.: 50-280
50-281
License No.: DPR-32
DPR-37

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to Surry Power Station Units 1 and 2.

Report No. 50-281/2003-001-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Very truly yours,

Richard H. Blount,
Site Vice President
Surry Power Station

Enclosure

Commitment contained in this letter:

A Root Cause Evaluation (RCE) was initiated to determine the cause of this event. The approved recommendations from the RCE necessary to prevent recurrence will be implemented through the corrective action program.

IE22

cc: United States Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23 T85
Atlanta, Georgia 30303-8931

Mr. R. A. Musser
NRC Senior Resident Inspector
Surry Power Station

NRC FORM 366 (7-2001)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB NO. 3150-0104 **EXPIRES 7-31-2004**

Estimated burden per response to comply with this mandatory information collection request 50 hours Reported lessons learned are incorporated into the licensing process and fed back to industry Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U S Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503 If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
SURRY POWER STATION , Unit 2	05000 - 281	1 OF 4

TITLE (4)

Electrical Conduit Bushing Failure Results in a Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER
01	25	2003	2003	-- 001 --	00	03	26	2003	SURRY POWER STATION, Unit 1	05000-280
									FACILITY NAME	DOCUMENT NUMBER 05000-

OPERATING MODE (9)	N	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
POWER LEVEL (10)	100 %		20.2201(b)		20.2203(a)(3)(ii)		50 73(a)(2)(ii)(B)		50 73(a)(2)(ix)(A)
			20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36©(1)(i)(A)	X	50 73(a)(2)(iv)(A)		73 71(a)(4)
			20 2203(a)(2)(i)		50.36©(1)(ii)(A)		50.73(a)(2)(v)(A)		73 71(a)(5)
			20.2203(a)(2)(ii)		50.36©(2)		50.73(a)(2)(v)(B)		OTHER
			20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
			20 2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
			20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
			20 2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
			20 2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Richard H. Blount, Site Vice President	(757) 365-2000

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
B	TB	JBX	W351	Y					

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

ABSTRACT (Limit to 1400 spaces, i e , approximately 15 single-spaced typewritten lines) (16)

On January 25, 2003 at 1414 hours with Unit 2 at 100% power, the reactor tripped due to turbine trip from a main generator trip. The main generator trip was due to Main Transformer and Generator Leads current differential lockout. The differential lockout resulted from a shorted main generator current transformer (CT) lead. Emergency systems functioned as required for the Unit 2 trip. At the time of the Unit 2 reactor trip, Unit 1 was off-line. Load shed occurred on the Unit 1 station service busses as designed. The Unit 1 'B' Main Feed Pump tripped, and since the Unit 1 'A' Main Feed Pump had previously been shut down, a start signal was initiated to both Unit 1 Motor Driven Auxiliary Feedwater Pumps (MDAFWPs) at approximately 1414 hours. The damaged CT leads were repaired. Inspection of like components on Units 1 and 2 was performed, and repairs made as necessary. The automatic actuation of the Unit 2 reactor trip, the actuation of Unit 2 Auxiliary Feedwater, and the automatic actuation of Unit 1 Auxiliary Feedwater are reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A).

1.0 DESCRIPTION OF THE EVENT

The Surry Main Generator Protection System provides fault protection for the main generator leads and main transformer by interrupting electrical flow if an electrical fault exists. A current transformer (CT) provides secondary current to a differential relay that provides this fault protection.

On January 25, 2003 at 1414 hours, with Surry Power Station Unit 2 at 100% reactor power, an automatic reactor trip was generated due to a main turbine generator trip. The main generator tripped due to a Main Transformer and Generator Leads differential lockout. All three auxiliary feedwater pumps automatically initiated as designed on low low steam generator level following the trip. All control rod bottom lights were lit, however, Individual Rod Position Indications (IRPIs) for three control rods indicated between 11 and 20 steps following the trip. In accordance with emergency operating procedures, emergency boration was initiated at 1434 hours and secured at 1440 hours, followed by normal boration to ensure adequate shutdown margin. The three IRPIs that initially did not indicate zero position drifted to less than 10 steps by 1501 hours. Boron concentration shutdown margin for Unit 2 was determined to be satisfactory at 1645 hours.

A load shed feature is provided to reduce electrical loading in the event two units were simultaneously loaded on the Reserve Station Service Transformers (RSSTs). This feature ensures that the voltages on the emergency busses will be within acceptable limits. At the time of the Unit 2 reactor trip, Surry Unit 1 was off-line and the Unit 1 electrical loads were provided by the RSSTs. When Unit 2 tripped, loads automatically transferred to the RSSTs and load shedding was initiated on Unit 1. As a result, the Unit 1 'B' Main Feed Pump tripped, and since the Unit 1 'A' Main Feed Pump had previously been shut down, a start signal was initiated to both Unit 1 Motor Driven Auxiliary Feedwater Pumps (MDAFWPs) at approximately 1414 hours. The Unit 1 'A' Main Feed Pump, was restarted at 1623 hours, and the Motor Driven Auxiliary Feedwater Pumps were secured at 1653 hours.

At 1725 hours, a four-hour and an eight-hour non-emergency report was made to the NRC as required by 10 CFR 50.72(b)(2)(iv)(B) and 10 CFR 50.72(b)(3)(iv)(A), respectively. The automatic actuation of the Unit 2 reactor protection system and the initiation of Unit 2 Auxiliary Feedwater (AFW) system are reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A). The automatic actuation of the Unit 1 AFW system is also reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A).

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event resulted in no significant safety consequences or implications. Emergency systems functioned as required for the Unit 2 trip. All three AFW pumps automatically initiated as designed on low low steam generator level following the trip. The operating

crew noted the three IRPI indications and in accordance with emergency procedures, initiated boration to ensure adequate shutdown margin. All electrical busses transferred properly following the trip and all emergency diesel generators were operable. The Reactor Coolant System (RCS) [EIIS-AB] cooled to a minimum average temperature (Tave) of approximately 540 degrees Fahrenheit (F) and then stabilized to the no load Tave value of 547 degrees F. No indications of primary to secondary leakage existed. Emergency systems functioned as required for the Unit 1 load shed. The actuation initiated flow from the two motor driven auxiliary feed pumps. In addition, the cross-connect from Unit 2 AFW system remained operable. Therefore, the health and safety of the public were not affected.

3.0 CAUSE

The Unit 2 automatic reactor trip was caused by a main generator trip. The main generator tripped due to a Main Transformer and Generator Leads differential lockout. It was observed by walkdown that a conduit fitting had separated from a junction box containing the wiring for the differential lockout CT secondary leads. A 'B' phase secondary lead became shorted when the associated conduit disconnected from the junction box [EIIS- TB, JBX] insulated bushing and the conduit became supported by the CT leads. The preliminary cause was the detachment of the insulated bushing locking collar allowing the insulated bushing and conduit connection to loosen, and ultimately disconnect. The inherent vibrations from the main turbine generator existing over a long period of time contributed to the failure.

The cause of the Unit 1 automatic start of the two MDAFWPs was the load shed feature designed to reduce electrical loading on the RSSTs. At the time of the Unit 2 reactor trip, Unit 1 was off-line. Load shedding tripped the operating Unit 1 Main Feed Pump and initiated a start signal for the Unit 1 MDAFWPs.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

The CT secondary leads were repaired and tested satisfactorily. The insulating bushing and conduit were repaired. Other Unit 2 insulating bushings were examined and other conduits containing protective relay wiring were identified as having degraded bushings. The insulating bushings and wiring were repaired and tested satisfactorily.

5.0 ADDITIONAL CORRECTIVE ACTIONS

Similar protective relay conduits and insulated bushings on Unit 1 were inspected and one insulated bushing was identified as having a disconnected locking ring. The Unit 1 insulated bushing was still attached to the conduit and was providing adequate support and protection for the internal CT wires. The insulating bushing was repaired.

6.0 ACTIONS TO PREVENT RECURRENCE

A Root Cause Evaluation (RCE) was initiated to determine the cause of the reactor trip due to generator differential. Conclusions from the RCE will be evaluated and the approved recommendations from the RCE necessary to prevent recurrence will be implemented through the corrective action program.

7.0 SIMILAR EVENTS

None.

8.0 MANUFACTURER/MODEL NUMBER

The insulated bushing was a Seimens Westinghouse Manufacturing part number 57D2226G01.

9.0 ADDITIONAL INFORMATION

None.